MULTIPURPOSED SMALL FAST REACTOR SVBR-75/100 COOLED BY PLUMBUM-BISMUTH

A.V. ZRODNIKOV, V.I. CHITAYKIN, B.F. GROMOV, O.G. GRIGORIEV,
A.V. DEDOUL, G.I. TOSHINSKY

STATE SCIENTIFIC CENTER OF RUSSIAN FEDERATION (SSC RF)
INSTITUTE OF PHYSICS AND POWER ENGINEERING (IPPE)
OBNINSK, RUSSIAN FEDERATION

YU.G. DRAGUNOV, V.S. STEPANOV

"HIDROPRESS" PILOT DESIGN BUREAU
PODOLSK, RUSSIAN FEDERATION

Abstract

Currently the nuclear power (NP) development meets significant difficulties in many countries. First of all it relates to complicating and cost rising of nuclear power plants (NPPs) due to essential enhancing the safety requirements. The possibility and expediency of developing the NP based on unified small power reactor modules SVBR-75/100 with fast neutron reactors cooled by lead-bismuth eutectic alloy is substantiated for the nearest decades in the Paper. Based on those modules the following designs can be realized: renovating of the NPP units which operation term has been exhausted; regional nuclear heat power plants of 100 - 300 MW power which need near cities' location; large power modular NPPs (~1000 MW) like US concept PRISM or Japanese concept 4S; nuclear power complexes for sea water desalinating in developing countries which meet non-proliferation requirements, reactors for plutonium utilization and minor actinides transmutation.

1. INTRODUCTION

Today, despite phasing out a lot of programs from nuclear power (NP), the interest in the nuclear technology based on using lead-bismuth liquid metal coolant for reactor cooling is enhanced. In our country this technology has been developed during several decades. Eight nuclear submarines (NS) with reactors cooled by lead-bismuth coolant (LBC) were constructed. The total operation time is about 80 reactor-years. The innovative nuclear power technology which has no analogs in the world has been demonstrated in our industry [1].

The interest in this technology is explained by the fact that the coolant natural properties enable to design the reactor installation (RI) with very high safety level. Besides, the simultaneous improving of technical and economical characteristics can be expected. Now the conditions for implementing this technology into civilian NP have been arisen.

2. SUBSTANTIATION OF CHOOSING THE FAST REACTOR COOLED BY LBC

Liquid metal cooled fast reactors are classified as RIs which safety is ensured mainly due to their inherent safety. It is associated with a number of their internal features.

Lack of poisoning effects in the fast reactor (FR), low value of negative temperature reactivity coefficient, compensation of fuel burn-up and slagging processes by plutonium generation enable to ensure the operative reactivity margin to be less than delayed neutron share and to eliminate the prompt neutrons runaway in the reactor under operation.

Liquid metal coolant (LMC) used for FR cooling considerably motivates the RI design and hydraulic scheme, the NPP technical and economical characteristics. Among the LMCs used in NP sodium is the most commonly used.

Choosing this LMC for fast reactors was caused by its opportunity of intensive heat removal due to its good thermal and physical properties. It enabled to provide short plutonium doubling time that was the obligatory requirement at the early phases of designing the fast breeder reactors in the 60s and 70s years, and was caused by the fact of unproved forecast of NP's very high development rate and, therefore, need for fuel self-providing. That was the reason why in the 50s Academician A.I. Leypunsky, who considered various LMCs for cooling the FRs, preferred sodium, though LBC was initially considered for these purposes [2].

Today and in the foreseeable future there is no need for such short plutonium doubling time that can maintain the sodium cooled FRs. The necessity for designing fast breeder reactors has been postponed to many decades [3]. It enables to use the opportunity of using LBC for FR cooling.

It should be highlighted that the experience of using sodium coolant has been gained in conditions of NPPs' power reactors industrial operation and could be used immediately, whereas the experience of using LBC has been acquired in conditions of NSs' RIs operation, which were different from those at NPP in design and operation regimes, and so this experience requires applicable adaptation to new conditions. However, these circumstances should not be the reason for not using LBC in NP due to weighty backgrounds.

These backgrounds are as follows:

- Enhancing the reactor safety due to:
  (1) Negative total void reactivity effect which is typical for small power reactors of SVBR-75/100 type and elimination of the possibility of realizing the local positive void reactivity effect. The last relates to impossibility of coolant boiling up in the most heat stressed fuel subassemblies (FSA) even in cases of severest accidents (LBC boiling point is ~ 1670^{o}C, sodium boiling point is ~ 870^{o}C);
  (2) Use of chemically inert LBC which eliminates arising of explosions and fires if the coolant comes in contact with water and air which is possible in emergency situations;
- Improving technical and economical parameters due to using two-circuit scheme of heat removal, eliminating some safety systems, systems of accident localization, simplifying the technology of managing the spent nuclear fuel (SNF);
- Solving the number of principal problems of LBC, which determine RIs reliability and safety: ensuring the proper coolant quality and its maintenance in the course of operation, ensuring radiation safety associated with forming alpha-active polonium-210 radionuclide, etc.;
- Closeness of RI SVBR-75/100 scale factor to that of NS's RI that enables to use many of technical solutions tried out by practice.

3. SAFETY ENSURING CONCEPT

High boiling point and latent evaporation heat, which are LBC natural properties, practically eliminate the possibility of the primary circuit over-pressurization and reactor thermal explosion at any considerable accidents because in this case no pressure increase arises.

Impossibility of coolant boiling up enhances the reliability of heat removal from the core and safety due to lacking the heat removal crisis phenomenon.

For proposed integral design of the RI the loss of coolant with its circulation interruption through the core caused by tightness loss of the reactor main vessel (postulated accident) is eliminated by introducing the safe-guard vessel, small free volume between the main reactor vessel and the safe-guard vessel and impossibility of coolant's boiling-up off in case of primary circuit gas system tightness loss.

In the case of failing all systems of emergency cooling down (postulated accident), elimination of core melting down under heat decay effect and keeping intact the vessel of small and medium-sized power reactors are ensured completely by passive way with large margin to boiling due to heat accumulation in internal reactor structures and in coolant with short-time increase of its temperature. In this case heat is removed through the reactor vessel (which temperature increases correspondingly) and the air gap to the water storage tank around the reactor vessel and into the ambient air at its natural circulation after boiling away the water (in case of full de-energizing and stopping the operation of cooling down systems for five days or more).

In the case of emergency overheating and simultaneous postulated failure of emergency protection systems (EPS), the reactor power decrease down to the level which does not cause the core damage is ensured by reactivity negative feedbacks.

The coolant itself reacts with water and air very slightly. The emergency processes caused by primary circuit pipes tightness loss and steam generator (SG) inter-circuit leaks occur without hydrogen release and any exothermic reactions. Within the core and RI there are no materials releasing hydrogen as a result of thermal and radiation effects and chemical reactions with coolant. Thus, the possibility of arising chemical explosions and fires caused by internal reasons is eliminated completely.

The elimination of water or steam penetration into the core caused by full rupture of SG tube and consequent over-pressurizing the reactor vessel designed for maximal possible pressure for that accident are ensured by coolant circulation scheme. In this scheme steam bubbles and water drops are thrown out on the free coolant level by upgoing coolant flow. Thereby steam effective separation occurs in the gas space of the primary circuit above the coolant level, whence steam goes to the passively operating emergency condensers system, and in case of their postulated failure or simultaneous rupture of several SG tubes it goes to the bubbler through the rupture membranes.

The operation experience of the RI using LBC at the NS has revealed the possibility of RI safety operation during some time in conditions of small SG leakage, which does not cause any significant deviations of the designed technological parameters. This fact allows to realize necessary repair works not urgently but at the convenient time.

Chemical inertness and impossibility of coolant's boiling up in case of the primary circuit tightness loss and its property to retain iodine, which radionuclides, as a rule, represent the major factor of radiation risk just after the accident, as well as the other fission products (inert gases are an exception) and actinides, reduce sharply the scale of radiation consequences of that accident.

The containment above the reactor serves as the additional safety system barrier. Its main purpose is the protection against external effects. Low storage of potential energy in the primary circuit restricts the RI destruction scale caused by over-standard external effects to only external impact forces.

As computations reveal, extremely high safety potential peculiar to this type of RIs is characterized by the fact that even when such initial events as containment destruction and primary circuit tightness loss coincide (that is possible in case of diversion or military attack), neither reactor runaway, nor explosion and fire occur, and the radioactivity exhaust is lower than that which requires the population evacuation. Taking into account that energy stored in the coolant (heating, chemical and compression potential energy) is minimal in comparison with other coolants used and previously mentioned physical special characteristics of fast reactors and RI integral design, one could look forward to designing the RI of extremely high safety level. On the base of those reactors NP would become not only socially acceptable for population but also socially attractive if it gained economical competitiveness with heat electric power plants using organic fuel (we have all backgrounds for it).

4. MULTI-PURPOSED REACTOR MODULE SVBR-75/100

RI SVBR-75/100 is designed for generating steam which parameters enable to use it as working medium in thermo-dynamical cycle of turbogenerator installations. It is possible to vary the steam parameters according to the needs. Today the base variant of RI SVBR-75 has been developed for generating the saturated steam under the pressure of 3.24 MPa, i.e., the pressure which is produced by the SG of the Novovoronezh NPP (NVNPP) second unit, and when turbogenerator with intermediate steam superheating is used, this enables to generate electric power of about 75 MWe when working under condensation regime.

The design of RI SVBR-75 module has two-circuit scheme of LBC heat removal for the primary circuit and steam-water for the secondary circuit. The integral design of the pool type is used for the RI primary circuit (see Fig. 1). It enables to mount the primary circuit equipment inside the one vessel. RI SVBR-75 module includes the removable part with the core (the reactor itself with control rods), 12 SG modules with compulsory circulation over the primary circuit and natural circulation over the secondary circuit, 2 main circulation pumps (MCP) for LBC circulation over the primary circuit, devices for controlling the LBC quality, the in-vessel radiation shielding and buffer chamber (pressurizer) which are the parts of the main circulation circuit (MCC).

The scheme of coolant moving within the MCC is as follows: through the windows of reactor outlet chamber the coolant heated in the core flows to the inlet of the SG twelve modules which have parallel connection. It flows from top to bottom in the intertube space of the SG modules and is cooled there. Then the coolant penetrates into the intermediate chamber, from which it moves in the channels of in-vessel radiation shielding, cooling it, to the reactor vessel upper part and there it forms the free level of “cold” coolant (peripheral buffer chamber), further from the reactor vessel upper part the coolant flow moves to the MCP suction inlet.

The adopted circulation scheme with free levels of LBC existing in the reactor vessel upper part and SG module channels, which contact the cover gas medium, ensures the reliable separation of steam-water mixture out of coolant flow when the accidental tightness loss of SG tube system occurs, and existing of gas medium ensures the possibility of coolant’s temperature changes.

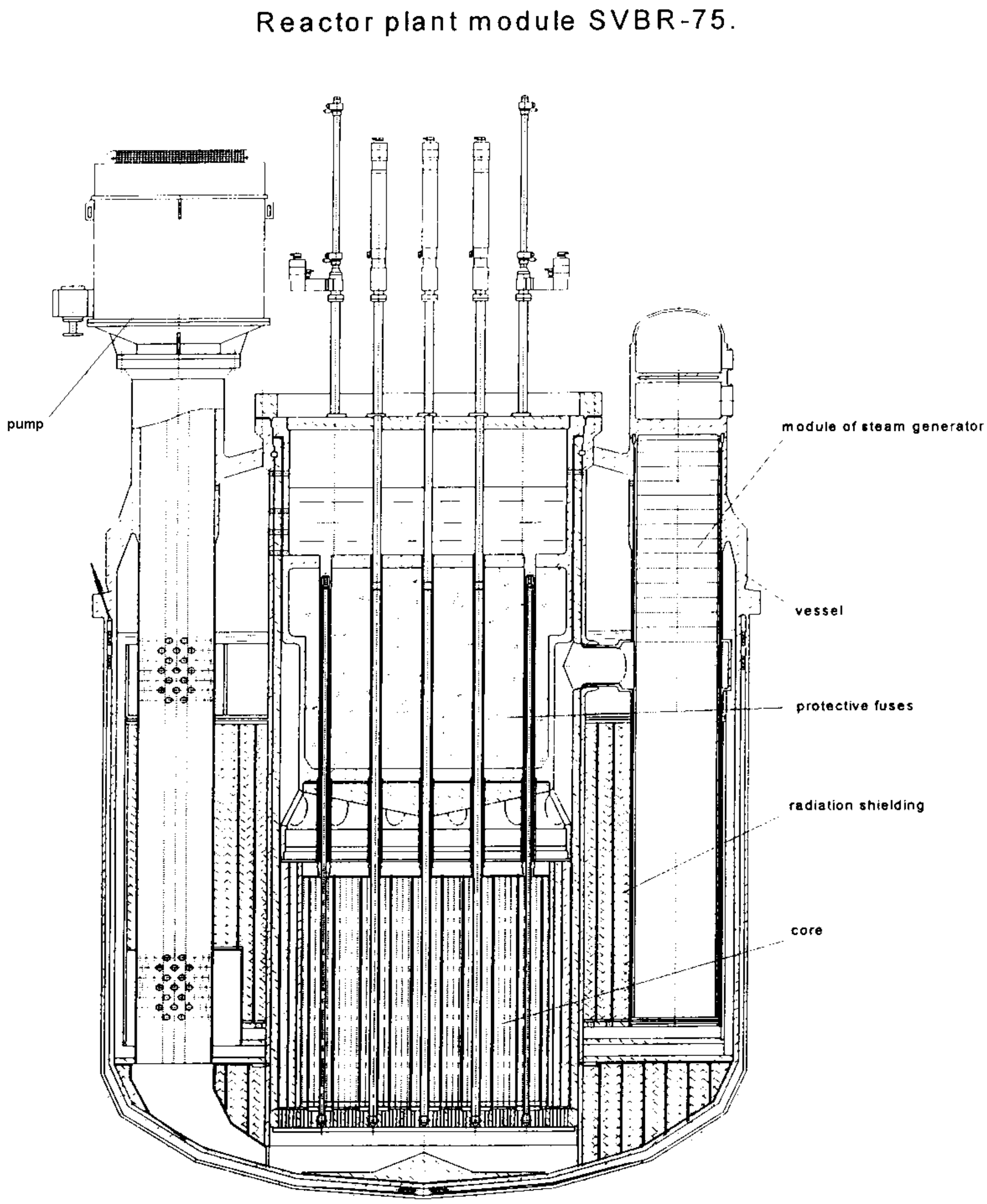


FIG. 1. Reactor general view.

Reactor vessel is placed in the tank and is mounted there (see Fig. 2). The tank is filled by water and is designed for cooling the RI in case of beyond design accidents. The gap between the main vessel and safe-guarding one is chosen to ensure the circulation circuit disrupture in case of accidents related to the tightness loss by the reactor vessel major vessel.

The secondary system is designed to operate the steam generator producing saturated steam with multiple natural circulation through the evaporator-separator circuit, as well as to provide the scheduled and emergency RI cooling by using steam generator.

The design provides three systems of heat removal to the heat sink both in scheduled and emergency reactor core cooling.

First system includes normal operation RI and turbine equipment and systems. The system is cooled by the primary heat removal via steam generator heat exchange surfaces, steam being dumped to the turbine generator systems (TGS).

The second heat removal system is an independent cooling system (ICS), which includes, besides a part of primary and secondary circuit equipment, a separator condenser (circuit) with natural circulation. Via this circuit the heat is removed to the intermediate circuit water. This system ensures independent (from the turbine generator systems) reactor cooling and independent reactor plant operation at a power level up to 6% N_{nom} at the nominal steam pressure. Connection/disconnection of ICS is realized with no operator action and without using external power supply systems.

Third core heat removal system is a passive heat removal system (PHRS). The heat is removed from the reactor to the water storage tank located around the reactor vessel. This system ensures the reactor core cooling in case of postulated maximal accident with all secondary circuit equipment failed, reactor protection system failure and total de-energizing of the NPP.

The principal technical parameters of RI SVBR-75 are presented in Table I.

RI SVBR-75 operates for eight years without core refueling. During this period there is no need in carrying out fuel works. At the initial stage the use of mastered oxide uranium fuel in the uranium closed fuel cycle is provided similar to that in reactor BN-600. Further the use of dense uranium and plutonium nitride fuel is possible. In this case the core breeding ratio is more than 1 and in the plutonium closed fuel cycle the reactor would operate by using only depleted waste pile uranium.

The refueling is performed after lifetime ending. Refueling means the complex of works on restoring the reactor full power resources which includes core replacing works, as well as dismantling and mantling works associated with them. FSA by FSA fuel unloading out of the reactor vessel is provided and loading of the fresh core as a part of new removable part is provided as well. Core refueling is realized by using special refueling equipment. Unloaded FSAs are placed in special capsules with liquid lead which solidifies further.

RI SVBR-75/100 is designed on the construction base of RI SVBR-75 and distinguishes from it only by SG operating in one through regime and generating the superheated steam of 400°C temperature and 9 MPa pressure. Thus the electric power is ensured to be of about 100 MWe.

Reactor plant SVBR-75.

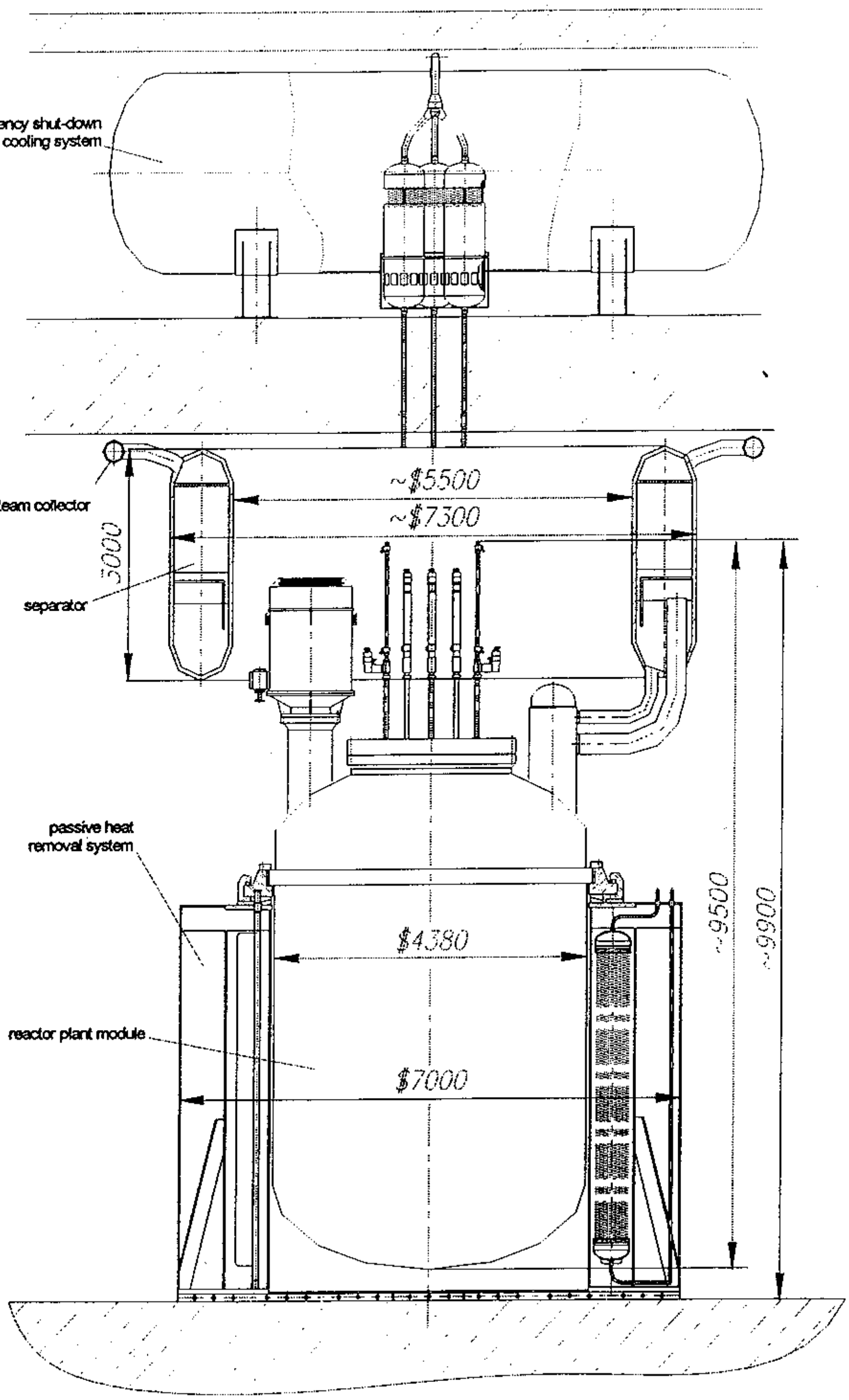


FIG. 2. Reactor plant SVBR-75.

TABLE I. PRINCIPAL TECHNICAL PARAMETERS OF RI SVBR-75

Parameter	Value
Number of reactors	1
Rated heat power, MW	268
Electric power, Mwe	75
Steam production rate, t/h	About 487
Steam parameters:	
- Pressure, Mpa	3.24
- Temperature	238
Feed water temperature, ºC	192
Primary coolant flow rate, kg/s	11180
Primary coolant temperature, ºC	
- Core outlet	439
- Core inlet	275
Core dimensions, D × H, m	1.65 x 0.9
Average value of specific volume core power, kW/dm^3	135
Average value of specific linear core power, kW/m	~ 22
Fuel:	
- Type	UO_2
- U-235 mass loading, kg	1476
- Average U-235 enrichment, %	15.6
SG numbers	2
Evaporator numbers in SG	6
Evaporator dimensions D × H, m	~ 0.6 × 4
Numbers of MCPs	2
MCP electric driver power, Kw	400
MCP head, Mpa	~ 0.5
Primary circuit coolant volume, m^3	18
Major reactor vessel dimensions, D × H, m	4.53 × 6.92
Designed earthquake of magnitude (MSK)	9
Designed construction terms, months	36

5. AREAS OF USING MULTIPURPOSED REACTOR MODULE SVBR-75/100

5.1. Renovation of NPP's units with exhausted lifetime

The number of NPP's units which lifetime has been exhausted is growing in the NP of many countries. It needs for huge expenditures on decommissioning the units out of operation and constructing the new ones for replacing the removing power capacities. At the same time there is an opportunity for untraditional solving this task by using NPP's units renovating. Renovating means replacing the RIs for units with exhausted lifetime by new RIs, using the NPP's existing buildings and structures with full replacing of removed power capacity. However, this way of replacing removed power capacities rigidly restricts the type of the RI used for renovation:

- The possibility of mountling the RI in existing rooms after dismantling the equipment of the “old” RI (the reactor which dismantling is followed by great radiation doses is an exception);
- Satisfying the regulation requirements for safety including “old” units without containment.

That way of replacing the removing power capacities of the second, third and fourth units of Novovoronezh NPP (NVNPP) based on the unique Russian lead-bismuth technology has been proposed by SSC RF IPPE together with “Hidropress” Design Bureau and “Atomenergoproekt” State Planning and Design, Research and Survey Institute and has been developed according to the task of concern “Rosenergoatom”.

SVBR-75 nominal power is chosen to be 75 MWe due to limited dimensions of NVNPP renovation units’ SG compartments which do not enable to install the large power module, necessity for ensuring the equality of generated steam and feeding water flow rates for SVBR-75 and RI VVER-440 SG, possibility of reactor module complete plant fabrication and its transportation by the railway, as well as closeness of the scale factor to NS’s RIs that enables to use some developed technical solutions and reduce R&D. For replacing the power capacities of the second unit four SVBR-75 modules are installed in SG compartments, and six modules are installed for each of third and fourth units.

Arrangement of modules in SG compartments of the MCP of NVNPP’s second unit is presented in Fig. 3.

The great economical efficiency of renovating the "old" NVNPP units is expected: specific capital renovation cost is $560 per kWe, that is half as many as that for constructing the new NPP unit [5].

5.2. Regional NHPP based on module SVBR-75/100

In the world many regions and first of all medium-sized and large cities face with serious difficulties in energy supplying especially heat supplying in winter.

The way for solving this problem is use of nuclear heating power plants (NHPP). However, use of traditional type RIs (which use water under the high pressure for reactor cooling) for these purposes needs designing the number of additional safety systems if compared with those accepted for NPPs situated at the distance of 25 km or more from the cities. This results in going up the NHPP cost and at the same time does not eliminate the principal possibility of scarcely probable nuclear accident but with severe consequences because the high pressure in the reactor, which is the internal cause of its arising, is not eliminated.

The high level of SVBR-75/100 reactor module inherent safety makes it possible and expedient to use it simultaneously for producing electric energy at the NPP and for heat generating at the NHPP which needs near city’s location. So we eliminate the possibility of arising severe accidents accompanied by explosions, fires with prohibited radioactivity exhaust requiring population evacuation beyond the NHPP site not only if there are personnel’s errors and equipment failures but if they coincide, if there are terrorist groups’ actions. It is possible to construct NHPPs of 100 - 300 MWe by using these standard modules.

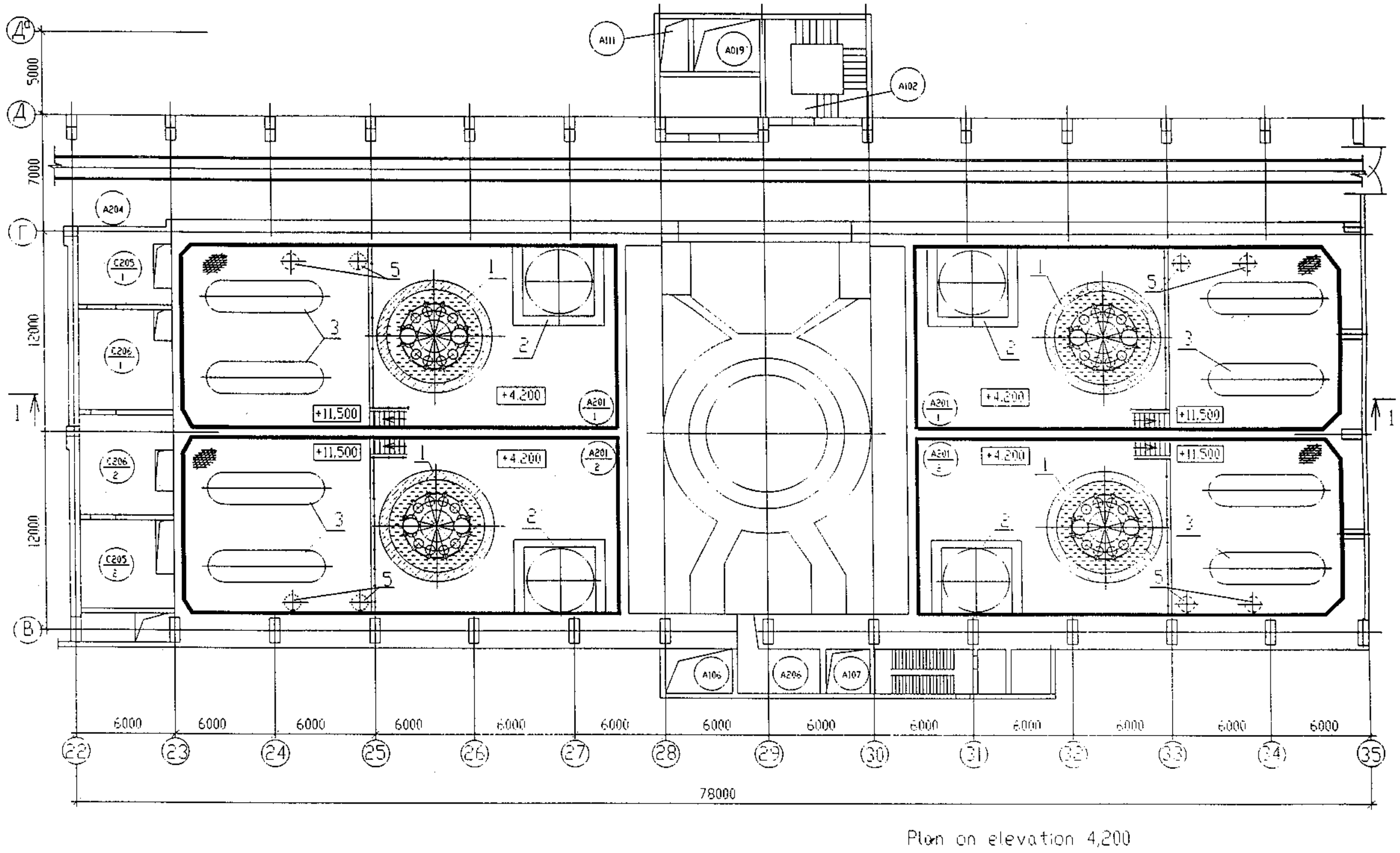


FIG. 3. SVBR-75 reactor modules arrangement at the NVNPP's second unit.

For RI SVBR-75/100 the principal design solutions for the unit's main building of regional NHPP distinguish from those for traditional type reactors. Small dimensions of SVBR-75/100 module and developed properties of inherent safety of FRs with LBC require the RI protection from only those external effects: aircraft falling, shock waves, maximal earthquake. There is no need for designing the tight shell withstanding significant internal pressure. Small dimensions of protected reactor compartment and simple scheme of RI enable to reduce the terms of NHPP unit construction and significantly reduce the construction cost.

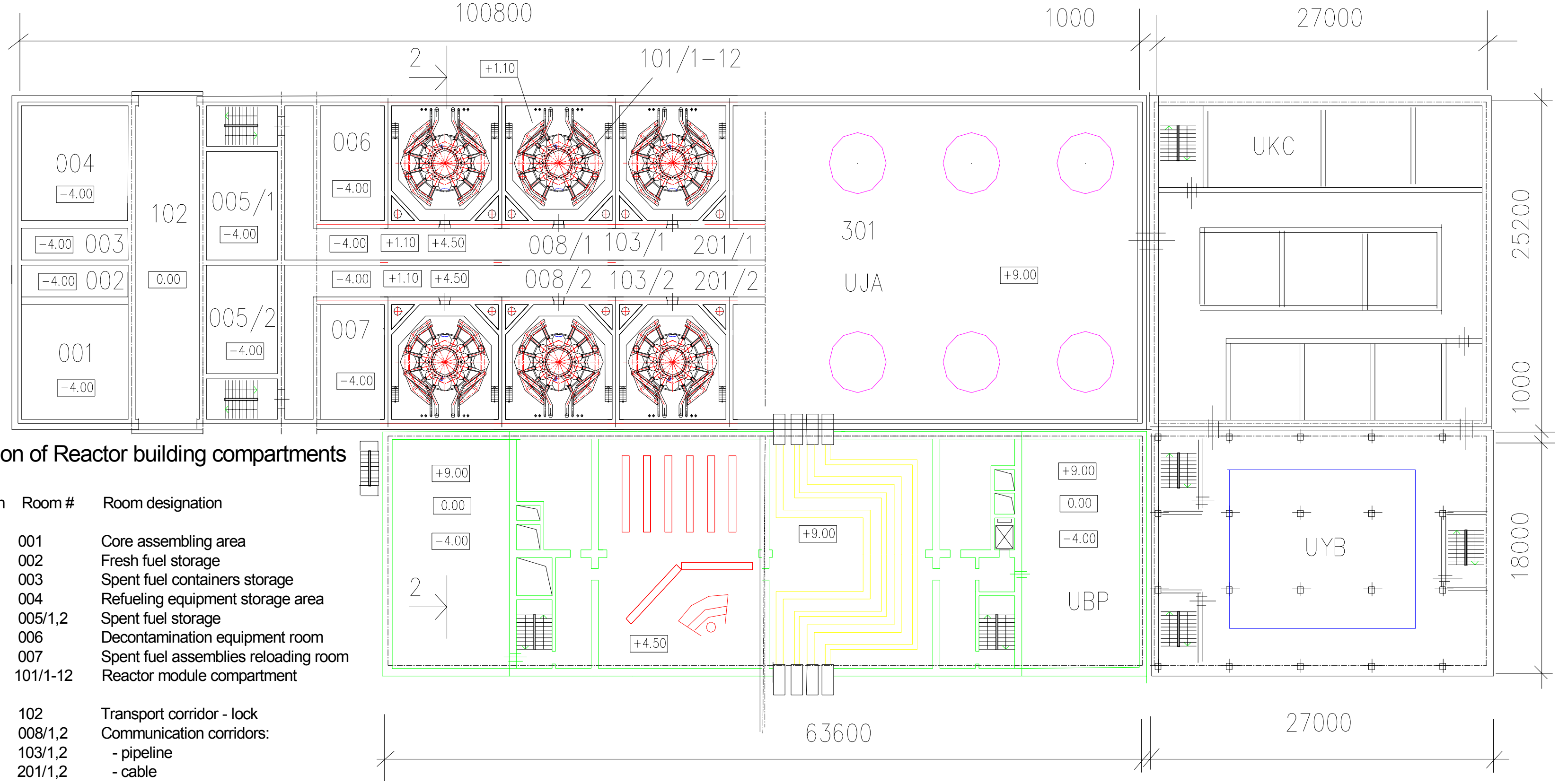
The simplicity of automated control system (ACS) conditioned by using passive systems for cooling down the RI facilitates the construction cost reducing.

As it has been assessed by experts, the economical competitive ability with HPP using fossil fuel will be ensured due to the following facts: almost lack of expenditures on nuclear fuel transportation; long lifetime of the fast reactor core, which ensures without refueling RI operation during about 8 years; low cost of spent nuclear fuel storing; almost lack of liquid radioactive waste and expenditures for its conditioning; complete plant fabricating of the RI and possibility of its transportation by truck, railway or sea to the NHPP constructing site that reduces the constructing terms and approaches them to those of traditional HPPs and reduces the investment cycle; sharp reduction of expenditures for constructing safety ensuring systems due to RIs' high inherent safety; high commercial production due to great demand for these NHPPs; possibility of export delivery of these RIs. The cost of these NHPPs is one-fifth to one-tenth as many as that of large NPPs.

5.3. Large power modular NPP (Figs 4 and 5)

On the basis of commercially produced modules SVBR-75/100 it is expedient to develop the design of modular NPP of large power (1 GWe and more at one unit). The prospect for that principle of designing NPP is shown in conceptual design developed in the USA (PRISM) [6] and in Japan (4S) [7]. However, use of this principle for LBC cooled reactors is the most effective.

The economical gain is achieved due to: constructing bulks reduction because of eliminating the number of safety systems and localizing systems, reducing the specific material expenditure (including bismuth demands) as compared to traditional reactors of large power, reducing the fabricating cost due to high commercial production, reducing the NPP constructing terms when reactor modules are delivered to the constructing site in high plant readiness. It enables to improve the conditions of credit receiving and repayment and to increase the competitive ability of NPP. The preliminary estimations have revealed that the specific capital cost of constructing that NPP is expected to be not more than that for NPP's VER-1000 unit.



Explication of Reactor building compartments

Elevation	Room #	Room designation
-4.00	001	Core assembling area
	002	Fresh fuel storage
	003	Spent fuel containers storage
	004	Refueling equipment storage area
	005/1,2	Spent fuel storage
	006	Decontamination equipment room
	007	Spent fuel assemblies reloading room
-4.00	101/1-12	Reactor module compartment
+1.10		
0.00	102	Transport corridor - lock
-4.00	008/1,2	Communication corridors:
+1.10	103/1,2	- pipeline
+4.50	201/1,2	- cable
		- blowing and exhaust venilation
+9.00	301	Reactor hall

FIG. 4. Nuclear island plan.

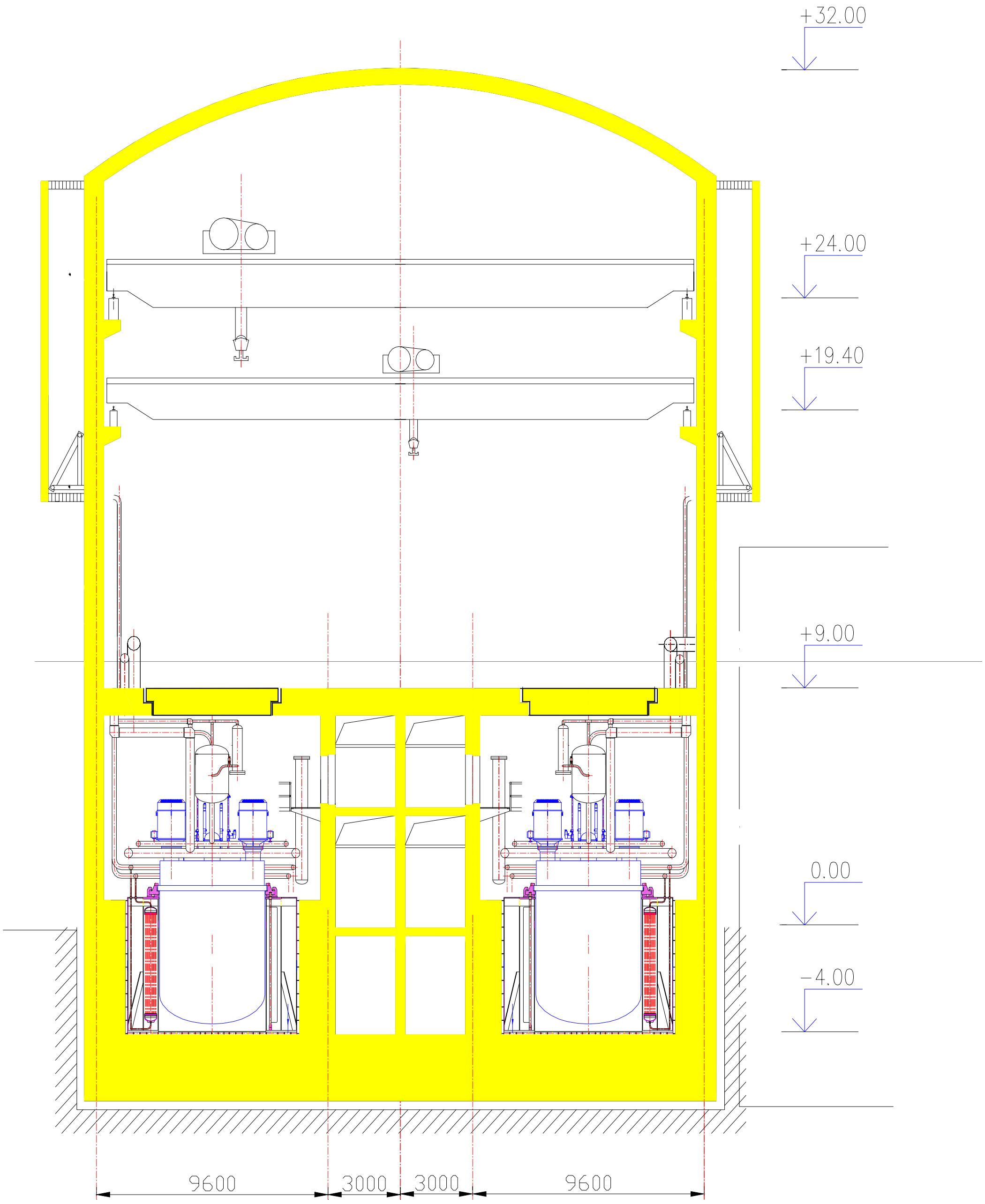


FIG. 5. Reactor compartment's sectional view.

5.4. Dual-purpose nuclear desalinating power complex for developing countries

Many developing countries in Africa and Asia suffer from deficiency of fresh water and electric energy. The majority of these countries do not have sufficient own resources of fossil fuel, which can meet their demands. In some countries fuel transporting is difficult, there are no powerful electric power transmission lines. The marketing researches conducted recently by IAEA [8] have revealed that in many cases small sized nuclear power sources of 100 MWe can be used economical effectively for these purposes.

However, the developing countries' particularities concerned with not sufficiently high level of education, technical culture, social and economical development, as well as possibilities of arising local military conflicts, put forward special requirements to the nuclear power technology which are stricter than those for developed countries.

First of all, these requirements are RI inherent safety against severe accidents that is based on RI inherent properties ensuring safety not only in cases of personnel's errors and multiple failures of technical systems coincidences, but in cases of sabotage terrorist actions, etc. Besides, they must meet strict non-proliferation requirements [9], including that refueling in the country-user must be eliminated and due to this fact the lifetime duration must be 10 years or more. The opportunity for reactor unit transportation to the country-manufacturer in the condition of nuclear and radiation safety for refueling and then transporting it to the country-user again must be ensured. Thefts of fuel must be technically eliminated as well. Besides, the competitive ability to the alternative resources of receiving fresh water and electric energy must be ensured.

RI SVBR-75/100 (Fig. 6) meets these requirements the most completely. It has extremely high safety potential, lifetime duration needed, ensures the regime of non-proliferation due to the following:

- Use of uranium with enrichment less than 20%;
- Lack of refueling in the country-user;
- Opportunity of transporting the reactor module after ending its lifetime to the country-manufacturer in the condition of nuclear and radiation safety with LBC "frozen" in the reactor.

5.5. Plutonium utilization and long-lived minor actinides transmutation

In different countries the policy on managing plutonium, the quantity of which increases steadily, is different. It is determined by the fact that on the one hand Pu is very valuable fuel for the future NP using FRs. On the other hand it can be used for political and military purposes. Besides, Pu is high radiotoxic material and is considered to be NP dangerous radioactivity waste. Long-lived minor actinides belong to it, too.

For the first case the issue of Pu managing (both extracted one and that is contained in spent nuclear fuel) results in long reliably controlled storing.

The second case results in the task of Pu transmuting into the form that reduces the risk of its unauthorized proliferation or its complete burning up and minor actinides nuclear transmutation.

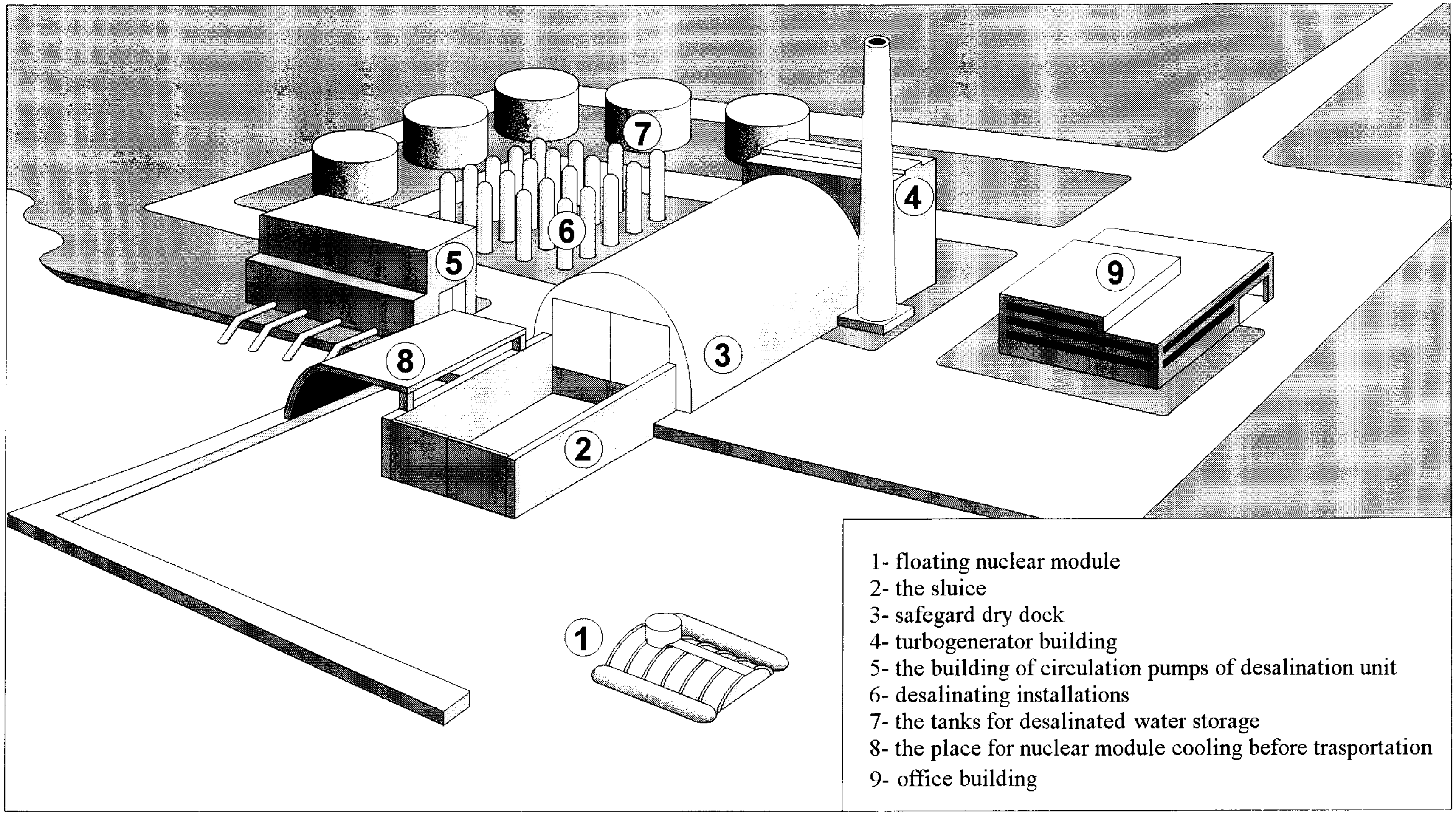


FIG. 6. General view of NPP site. Variant with electric supply and sea water desalination.

For solving this task the works on mastering the new nuclear technology using accelerator-driven systems are carried out. The main stimulus for developing this technology is Pu and minor actinides blanket subcriticality that eliminates the prompt neutrons runaway nuclear accident.

Along with it, this task can be solved on the basis of already mastered technology. For example, during eight years one reactor module SVBR-75/100 can transmute about 1000 kg of Pu (weapons- or reactor grade one) into the form protected against unauthorized proliferation ("spent fuel standard") at reducing its quality as a weapon material compared to the weapon Pu. In terms of 1 GWe·year 1.25 t of Pu will be utilized in those reactors. If minor actinides (first of all americium) is introduced into fuel, their transmutation into short-lived radioactive wastes will take place.

The safety level needed will be ensured due to developed propertied of RI SVBR-75/100 inherent safety which have been mentioned above.

6. THE PROBLEMS OF INCREASING THE OVERALL BISMUTH PRODUCTION AND EXPLOITING THE LEAD COOLANT

The factor limiting LBC use in large-scale future NP depends on deficiency of today's bismuth manufacturing, which has been determined by its consumption level.

Today's bismuth situation looks like uranium one during the period from 1906 to 1940 when only 4000 t of uranium were mined over the 34 years. But in 1980 the world uranium mining (except the ex-USSR) reached 40 000 t per year [10]. The explored uranium resources increased to the great extent too.

One should point out that bismuth content in the earth's crust is one-fifth as much as that of lead [11]. However, deposits of bismuth with high content of about 5 – 25% are very rare and locate in Bolivia, Tasmania, Peru and Spain. That is why 90% of world's bismuth has been manufactured from the wastes of lead-refining, copper-smelting and tinning plants.

The experience of the USA and Japan has revealed that equipping copper-smelting, tungsten and lead plants by dust-catching and dust-utilizing systems can essentially enhance bismuth extraction and has great significance for the environment.

In Russia that work can be launched at MINATOM enterprises being the members of AO "Atomredmetzoloto". According to VNIPI Promtechnologia information, the production of bismuth of about 2500 – 3000 t per year, together with gold and other metals, can be organized in the south-east of Chita Region where the resources of gold-bismuth ore have already been explored. That volume of bismuth production will ensure the year input of 2.5 - 3.0 GWe for the NPP using RI SVBR-75/100. It is necessary to carry out bismuth geological works, develop existing and new mines, implement the technology of deep bismuth extracting.

Though the cost of bismuth is ten times as much as that of lead it is a very small part of capital costs for NPP construction. It should be taken into account, that coolant is not spent and could be used again in other RIs.

RDIPE proposes to consider lead coolant [12] as an alternative to lead-bismuth alloy because the scales of lead production do not limit the rate of large-scale NP development.

However, use of lead coolant is associated with some engineering problems. Due to the higher lead melting point (it is 327°C against 125°C for eutectic alloy), lead coolant temperature must be increased significantly. It complicates the solution to the problems on coolant technology, structure materials corrosion resistance and mass transfer. Being applied to the lead-bismuth coolant, the problem has taken about 15 years for its solving. Besides that, it results in more complicated RI operation because of great opportunity of forming solid "sows" in the primary circuit under transitive regimes, accidents, repairs, refueling and not clear today the opportunity of safe for RIs melting the "sows". Nowadays the works on mastering the lead coolant are at their initial phase.

Taking into account all mentioned above, use of lead coolant would be justified only if the rates of power capacities increase for the NPPs with the RIs considered were high enough, and expenditures for increasing the annual bismuth production and its cost were put up as much that the increase of specific capital costs for NPP construction would not be economically reasonable.

As bismuth has been in deficiency, one can consider non-eutectic alloy with bismuth content decreased up to 10% (versus 56% in eutectic alloy). Being compared with lead coolant, its melting temperature is decreased by 77°C (to 250°C) and that facilitates RI operation and reduces maximal temperatures of fuel elements claddings up to the values tested for eutectic alloy under the conditions of long-term operation tests.

7. CONCLUSION

The experience gained in the course of designing the LBC cooled NS RIs enabled to design multi-purposed small size reactor module SVBR-75/100, which ensures the most complete realization of inherent safety principle for severest accidents and their deterministic elimination due to using the fast reactor, LBC and the primary circuit integral design. Based on these modules, renovation of NPP's units exhaust their lifetime can be carried out, regional NHPPs for energy shortage regions can be built, large power NPPs of modular type can be constructed, meeting IAEA requirements power complexes for electric energy producing and seawater desalinating can be built in developing and other countries, reactors for utilizing Pu and transmuting the long-lived minor actinides can be designed.

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